Exelon Generation Company, LLC
Braidwood Station
35100 South Rt 53, Suite 84
Braceville, IL 60407-9619
Tel. 815-458-2801

www.exeloncorp.com

Exelon
Nuclear

July 17, 2001
BW010082

U. S. Nuclear Regulator Commission
ATTN: Document Control Desk
Washington, DC 20555-0001

Braidwood Station, Unit 2
Facility Operating License No. NPF-77
NRC Docket No. STN 50-457

Subject: Submittal of Licensee Event Report Number 01-001-00

The enclosed Licensee Event Report (LER) is being submitted in accordance with 10 CFR 50.73(a)(2)(iv). 10 CFR 50.73(a) requires an LER to be submitted within 60 days after discovery of the event. Therefore, this report is being submitted by July 18, 2001.

Should you have any questions concerning this letter, please contact Amy Ferko, Regulatory Assurance Manager, at (815) 417-2699.

Respectfully,

James D. von Suskil
Site Vice President
Braidwood Station

Enclosure: LER Number 01-001-00

cc: Regional Administrator - Region III
NRC Braidwood Senior Resident Inspector

IE22

NRC FORM 366 (1-2001)	U.S. NUCLEAR REGULATORY COMMISSION	APPROVED BY OMB NO. 3150-0104	EXPIRES 6-30-2001

LICENSEE EVENT REPORT (LER)

FACILITY NAME (1)	DOCKET NUMBER (2)	PAGE (3)
Braidwood, Unit 2	STN 05000457	1 of 5

TITLE (4)
Braidwood Unit 2 Reactor Trip and Subsequent Loss of Non-Safety Related Offsite Power Due to Failure to Perform Concurrent Verification and Improper Command and Control

EVENT DATE (5)			LER NUMBER (6)			REPORT DATE (7)			OTHER FACILITIES INVOLVED (8)	
MO	DAY	YEAR	YEAR	SEQUENTIAL NUMBER	REV NO	MONTH	DAY	YEAR	FACILITY NAME	DOCKET NUMBER
									N/A	N/A
05	19	2001	2001	001	00	07	17	2001	FACILITY NAME N/A	DOCKET NUMBER N/A

OPERATING MODE (9)	1
POWER LEVEL (10)	100

THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: (Check one or more) (11)

☐ 20.2201(b)	☐ 20.2203(a)(3)(ii)	☐ 50.73(a)(2)(ii)(B)	☐ 50.73(a)(2)(ix)(A)
☐ 20.2201(d)	☐ 20.2203(a)(4)	☐ 50.73(a)(2)(iii)	☐ 50.73(a)(2)(x)
☐ 20.2203(a)(1)	☐ 50.36(c)(1)(i)(A)	■ 50.73(a)(2)(iv)(A)	☐ 73.71(a)(4)
☐ 20.2203(a)(2)(i)	☐ 50.36(c)(1)(ii)(A)	☐ 50.73(a)(2)(v)(A)	☐ 73.71(a)(5)
☐ 20.2203(a)(2)(ii)	☐ 50.36(c)(2)	☐ 50.73(a)(2)(v)(B)	☐ OTHER Specify in Abstract below or in NRC Form 366A
☐ 20.2203(a)(2)(iii)	☐ 50.46(a)(3)(ii)	☐ 50.73(a)(2)(v)(C)	
☐ 20.2203(a)(2)(iv)	☐ 50.73(a)(2)(i)(A)	☐ 50.73(a)(2)(v)(D)	
☐ 20.2203(a)(2)(v)	☐ 50.73(a)(2)(i)(B)	☐ 50.73(a)(2)(vii)	
☐ 20.2203(a)(2)(vi)	☐ 50.73(a)(2)(i)(C)	☐ 50.73(a)(2)(viii)(A)	
☐ 20.2203(a)(3)(i)	☐ 50.73(a)(2)(ii)(A)	☐ 50.73(a)(2)(viii)(B)	

LICENSEE CONTACT FOR THIS LER (12)

NAME	TELEPHONE NUMBER (Include Area Code)
Richard Graham, Operations Manager	(815) 417-2200

COMPLETE ONE LINE FOR EACH COMPONENT FAILURE DESCRIBED IN THIS REPORT (13)

CAUSE	SYSTEM	COMPONENT	MANU-FACTURER	REPORTABLE TO EPIX	CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO EPIX
N/A	N/A	N/A	N/A	N/A	N/A	N/A	N/A	N/A	N/A

SUPPLEMENTAL REPORT EXPECTED (14)			EXPECTED SUBMISSION DATE (15)	MONTH	DAY	YEAR
☐ YES (If yes, complete EXPECTED SUBMISSION DATE).	X	NO		N/A	N/A	N/A

ABSTRACT (Limit to 1400 spaces, i.e., approximately 15 single-spaced typewritten lines) **(16)**

At 0406 on 05/19/01, Braidwood Unit 2 tripped due to a loss of loop 2C reactor coolant flow while operating at 100 percent power. The 2C reactor coolant pump tripped when a non-licensed operator opened the incorrect system auxiliary transformer (SAT) feed potential transformer fuse door on 6.9kV Bus 258 during preparations for a Unit 2 SAT outage. The loss of the 2C reactor coolant pump resulted in an automatic reactor trip due to low flow in the 2C loop.

Due to the existing electrical lineup in support of the SAT outage, power to the 6.9kV buses and non-ESF 4kV buses was lost when the turbine/generator tripped offline. This resulted in a loss of all reactor coolant pumps, circulating water pumps, the main feedwater pumps and the condensate/condensate booster pumps. The reactor coolant system was maintained on natural circulation cooling with the secondary heat sink supplied by the steam generator power operated relief valves and auxiliary feedwater supplying the steam generators.

The root causes for the event were failure to perform concurrent verification to identify the correct SAT primary potential transformer fuse door and improper command and control to ensure critical tasks were performed correctly. The corrective actions are counseling and discipline of the responsible individuals and development and implementation of a policy to strengthen command and control standards for execution of critical activities in the plant.

Plant Operating Conditions Before The Event:

Unit: 2 Event Date: 5/19/2001 Event Time: 0406 hours

MODE: 1 Reactor Power: 100.0 percent

Reactor Coolant System [AB] Temperature: 580 degrees F, Pressure: 2235 psig

B. Description of Event:

The Unit 2 system auxiliary transformer [EB] was out of service at the beginning of the event. This is the only equipment that was inoperable at the beginning of the event that contributed to the event.

On the afternoon shift of Friday 5/18/01 with Unit 2 at full power, Operations was in the process of removing the Unit 2 system auxiliary transformers (SATs) from service. The 6.9kv buses (258 and 259) [EA] that are normally powered by the Unit 2 SATs were transferred to the Unit 2 unit auxiliary transformers (UATs) [EA] to support the SAT outage. The Unit 2 safety related 4kv ESF buses [EB] that are normally powered by the Unit 2 SAT's were transferred to the Unit 1 SATs.

On Saturday 05/19/01, the SAT outage Project Manager (PM), an off-shift Senior Reactor Operator (SRO), led the midnight shift operations crew in a pre-job briefing. The Shift Manager, the Unit 2 Supervisor, both Unit 2 Nuclear Station Operators (NSOs), and four extra non-licensed operators participated in the pre-job briefing. The PM stressed the importance of operating the correct potential transformer fuse doors during the pre-job briefing. The Field Supervisor (FS) and the Work Execution Center (WEC) Supervisor did not participate in the pre-job brief. The FS was in the field investigating a problem on Unit 1 at the time of the pre-job briefing. Shift supervision failed to recognize the need to have the FS present at the briefing. This contributed to the improper command and control of the SAT outage evolution.

The non-licensed operators assigned to the task had not performed this evolution previously and did not have experience working with potential transformer fuses. Because of the inexperience of the non-licensed operators performing this evolution, the PM directed the FS to observe the potential transformer fuse removal. Although the FS was not present at the pre-job briefing, the FS was experienced in this task and had observed performance of this task in the past.

When the non-licensed operators were ready to remove the first set of potential transformer fuses, they contacted the FS for further direction and guidance per the pre-job briefing instructions. The FS explained and supervised removal of the potential transformer fuses for bus 241 SAT feed potential transformer. Within the fuse cabinet, there is an upper and a lower door. One door is for the bus potential transformer, the other is for the SAT feed potential transformers. The potential transformer fuse removal on bus 241 and bus 242 required the upper

door labeled "SAT FEED POTENTIAL TRANSFORMERS" to be opened. Prior to beginning the potential transformer fuse removal for the SAT feed to bus 242, the Unit 1 Nuclear Station Operator (NSO) contacted the FS for a problem not related to the Unit 2 SAT outage. Before leaving the area, the FS questioned the non-licensed operators to ensure they were comfortable with continuing the fuse removal evolution in his absence. The non-licensed operators answered in the affirmative and the FS left the area. The non-licensed operators successfully removed the bus 242 potential transformer fuses. The non-licensed operators then proceeded to the 6.9kv switchgear room to continue with bus 258 potential transformer fuse removal.

The non-licensed operators identified the correct external cubicle door for 6.9kv bus 258 using concurrent verification. A third non-licensed operator already wearing protective gear from another evolution joined the group to complete the fuse removal evolution. This non-licensed operator had attended the pre-job briefing.

The non-licensed operator who had the procedure for the task directed the non-licensed operator in protective gear to open the upper door for the SAT feed to bus 258 primary potential transformer fuses. This direction was incorrect for the 6.9kv buses and was not what was stated in the procedure. The procedure stated to pull open the lower door labeled "SAT FEED POTENTIAL TRANSFORMERS" for 6.9kv bus 258. The labeling on the correct door exactly matched the procedure step that was in progress. Neither of the other two non-licensed operators read the procedure step. One assumed that the 6.9kv bus SAT feed potential transformer fuses would be contained in the same location as the previous two sets of 4kv bus SAT feed potential transformer fuses which had just been pulled. The non-licensed operator in protective gear had not been present for the previous fuse removal, and he assumed that the procedure had been read correctly.

The non-licensed operator in the protective gear, using 3-way communication, repeated back the instructions, pointed to the upper door and asked if the upper door was the correct component to manipulate. The non-licensed operator with the procedure in-hand replied that it was the correct door. The non-licensed operator in protective gear opened the upper door on verbal direction and caused an under voltage on bus 258 which tripped the 2C reactor coolant pump (RCP) [AB]. The three non-licensed operators heard the 2C RCP breaker open and observed that the indicating light had changed from red to green. At this time, the Unit 2 reactor automatically tripped due to loss of flow in the 2C reactor coolant loop with reactor power greater than 30%. All control rods inserted and all appropriate safety systems operated as designed.

The reactor trip caused a trip of the Unit 2 main turbine/generator [TB]. Due to the alignment of the non-safety related buses to the Unit 2 UATs and with the Unit 2 SATs de-energized, all non-safety related 4kv and 6.9kv buses de-energized. The remaining three RCPs tripped along with the circulating water pumps [SG], the main feedwater pumps [SJ] and the condensate/condensate booster pumps [SD]. The Unit 2 primary plant was placed on natural circulation cooling with the secondary heat removal from the steam generator power operated relief valves. The auxiliary feedwater [BA] pumps started on the RCP bus under voltage

signal and provided feed flow to the steam generators. The final result was a loss of all non-safety related off-site power to Unit 2. The Unit 2 safety related 4kv ESF buses were cross-tied to Unit 1 and remained energized throughout the event. The Unit 2 diesel generators [EK] were subsequently manually started and buses 241 and 242 were transferred to the 2A and 2B Diesel Generators.

C. **Cause of Event:**

The reactor tripped when the incorrect door was opened on bus 258 for the removal of bus 258 potential transformer fuses. Opening the incorrect door caused bus 258 to de-energize which caused the 2C RCP to trip. The trip of the 2C RCP caused the reactor to trip on loss of flow. The reactor trip caused the main generator to trip which then resulted in all non-safety related 4kv and 6.9kv busses de-energizing.

The first root cause for this event was the failure of non-licensed operators to perform concurrent verification of the SAT primary potential transformer fuse door.

The second root cause for this event was improper command and control of the critical steps within the evolution.

D. **Safety Consequences:**

The reactor protection system responded normally following a loss of flow to one reactor coolant loop, and the reactor was placed in a sub-critical state. Due to the existing electrical lineup, the event resulted in loss of forced circulation through the reactor coolant system and a loss of main condenser vacuum. Core cooling was maintained by operation on natural circulation with secondary heat sink provided from the steam generator power operated relief valves and auxiliary feedwater. Natural circulation is a design feature of the plant and is discussed in the Updated Final Safety Analysis Report (UFSAR). As discussed in the UFSAR, natural circulation flow in the RCS following a loss of forced coolant flow is sufficient to remove residual heat from the core.

There were no safety consequences as result of the event as no margins were exceeded. All safety related equipment required to mitigate the consequences of this event was available following the reactor trip, and operated as designed.

The event did not result in a Safety System Functional Failure.

E. **Corrective Actions:**

The first root cause for the event was the failure of the non-licensed operators to perform concurrent verification as required by station procedures for the SAT feed potential transformer fuse door. The corrective action to prevent recurrence of this root cause was counseling and personnel discipline in accordance with station policy. An additional corrective action was a station

stand down. During the stand down, the Site Vice President and Station Manager reviewed the event and the human error prevention aspects that failed during the evolution with station employees.

The second root cause for the event was improper command and control of the critical steps within the evolution. The corrective action to prevent recurrence of this root cause was the issuance of an Operations Department Standing Order, "Command and Control Standard for Execution of Critical Steps." This standard has been issued at all Exelon Nuclear sites. This standard states the requirements to screen all activities for risk impact, to identify critical steps that require direct oversight and to obtain authorization from the Main Control Room prior to performing critical steps. An additional corrective action was counseling and personnel discipline in accordance with station policy.

F. Previous Occurrences:

There have been no similar occurrences at Braidwood Station.

G. Component Failure Data:

Manufacturer	Nomenclature	Model	Mfg. Part Number
N/A	N/A	N/A	N/A